August 4, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000456/2006003; 05000457/2006003

Dear Mr. Crane:

On June 30, 2006, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Braidwood Station, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on June 30, 2006, with Mr. K. Polson and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one NRC-identified finding of very low safety significance (Green) is documented in this report. The issue was determined to involve a violation of NRC requirements. Because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a Non-Cited Violation in accordance with Section VI.A of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Braidwood facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Richard A. Skokowski, Chief
Branch 3
Division of Reactor Projects

Docket Nos. 50-456; 50-457
License Nos. NPF-72; NPF-77

Enclosure: Inspection Report 05000456/2006003; 05000457/2006003
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Braidwood Station
Plant Manager - Braidwood Station
Regulatory Assurance Manager - Braidwood Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing
Manager Licensing - Braidwood and Byron
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report No:	05000456/2006003; 05000457/2006003
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	April 1 through June 30, 2006
Inspectors:	S. Ray, Senior Resident Inspector G. Roach, Resident Inspector B. Bartlett, Senior Resident Inspector, Byron E. Bonano, Health Physicist C. Brown, Reactor Inspector J. Cassidy, Health Physicist J. Fuller, Reactor Inspector M. Holmberg, Reactor Inspector S. Orth, Senior Health Physicist B. Palagi, Operator License Examiner R. Ruiz, Reactor Engineer N. Shah, Project Engineer D. Smith, Project Engineer W. Snell, Senior Health Physicist B. Metro, Illinois Emergency Management Agency M. Perry, Illinois Emergency Management Agency J. Roman, Illinois Emergency Management Agency
Observers:	S. Ray, General Engineer D Lords, Reactor Inspector
Approved by:	R. Skokowski, Chief Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000456/2006003, 05000457/2006003; 04/01/2006 - 06/30/2006; Braidwood Station, Units 1 & 2; Fire Protection.

This report covers a 3-month period of baseline inspection, an inspection in accordance with Temporary Instruction (TI) 2515/150, "Reactor Pressure Vessel Head and Vessel head Penetration Nozzles," and a followup inspection of certain portions of TI 2515/165, "Operational Readiness of Offsite Power and Impact on Plant Risk." The inspections were conducted by resident and inspectors based in the NRC Region III office. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Non-Cited Violation of Braidwood Facility Operating License Nos. NPF-72 and NPF-77, Condition 2.E, for failing to maintain the firewall separating the fuel handling building and the auxiliary building in accordance with the approved fire protection program. Fire dampers were required to be provided in this firewall, except where an evaluation had been performed and approved to allow a deviation. Dampers were not installed in two ventilation ducts in the firewall separating the spent fuel pool heat exchanger rooms of the fuel handling building and the Unit 1 and Unit 2 containment pipe penetration areas of the auxiliary building; also, no evaluation or exemption existed to justify this configuration. The licensee entered the issue into its corrective action program for resolution, implemented compensatory measures that included hourly fire watches.

This finding was more than minor because it affected the Mitigating Systems Cornerstone objective to ensure that external factors (i.e., fire, flood, etc) do not impact the availability, reliability, and capability of systems that respond to initiating events. The finding was of very low safety significance because the steel ventilation duct provided a minimum of 60 minutes fire endurance protection and the location of combustibles were positioned such that the unprotected duct penetration would not be subjected to direct flame impingement. (Section 1R05)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 started a gradual power coastdown on April 3, 2006, and reached about 93 percent power on April 16, 2006, when the unit was taken off line and shutdown for a refueling outage. Unit 1 was brought critical and the generator was synchronized to the grid on May 3, 2006. Unit 1 reached full power on May 8, 2006, and operated at or near full power for the remainder of the inspection period except that power was briefly reduced to about 95 percent on June 16, 2006, at the request of Electric Operations due to grid conditions.

Unit 2 operated at or near full power throughout the inspection period except that power was briefly reduced to about 97 percent on April 6, 2006, in order to isolate a failed open feedwater system relief valve and power was briefly reduced to about 95 percent on June 16, 2006, at the request of Electric Operations due to grid conditions.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Readiness for Seasonal Susceptibilities

a. Inspection Scope

The inspectors reviewed the licensee's seasonal preparations for operation during the summer months. This was primarily accomplished by verifying that the licensee had completed the requirements for summer readiness as documented in Exelon Nuclear Procedure WC-AA-107, "Seasonal Readiness." The inspectors also reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS) and other design-bases documents to identify those components that were susceptible to degradation from high temperatures during the summer months. The inspectors verified that the licensee had addressed these components in preparation for summer operation. In addition, the inspectors selected the following risk-significant support systems/areas for specific review:

- Units 1 and 2 main power transformers and bus duct cooling; and
- auxiliary building chiller reliability.

The inspectors also reviewed several issue reports (IRs) documenting problems with bus duct fan preventive maintenance lessons learned, rescheduling of an action tracking item affecting summer readiness, and seasonal readiness peer review results to determine whether these issues were being properly addressed in the licensee's corrective action program. In addition, the inspectors reviewed the licensee's common cause analysis, "Summer Readiness Issues," and a completed work order on high temperature equipment protection." The inspectors verified that minor issues identified

during these inspections were entered into the licensee's corrective action program. Documents reviewed in this inspection are listed in the Attachment.

This review constituted two samples of this inspection requirement.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of the accessible portions of risk-significant system trains during periods when the train was of increased importance due to redundant trains or other equipment being unavailable. The inspectors utilized the valve and electric breaker listed to determine whether the components were properly positioned and that support systems were aligned as needed. The inspectors also examined the material condition of the components and observed operating parameters of equipment to determine whether there were any obvious deficiencies. The inspectors reviewed IRs associated with the train to determine whether those documents identified issues affecting train function. The inspectors used the information in the appropriate sections of the TS and the UFSAR to determine the functional requirements of the system. The inspectors also reviewed the licensee's identification of and the controls over the redundant risk-related equipment required to remain in service. Documents reviewed during this inspection are listed in the Attachment.

The inspectors completed three samples of this requirement by walkdowns of the following trains:

- 1A residual heat removal (RH) train electrical and mechanical line-up prior to 1B RH train outage;
- 2A diesel generator (DG) electrical and mechanical line-up prior to 2B DG outage; and
- 2B containment spray (CS) train electrical and mechanical line-up prior to 2A CS train outage.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors conducted fire protection walkdowns that focused on the availability, accessibility, and condition of fire fighting equipment, on the control of transient combustibles and ignition sources, and on the condition and operating status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events, with additional insights on their potential to impact equipment which could initiate a plant transient or be required for safe shutdown. The inspectors used the Fire Protection Report, Revision 21, to determine: that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition.

The inspectors completed nine samples of this inspection requirement during the following walkdowns:

- fuel handling building (Fire Zone 12.1-0);
- Unit 2 non-segregated bus-duct area (Fire Zone 3.2a-2);
- Unit 2 Division 21 engineered safety feature (ESF) switchgear room (Fire Zone 5.2-2);
- Unit 2 Division 22 ESF switchgear room (Fire Zone 5.1-2);
- Unit 1 Division 11 ESF switchgear room (Fire Zone 5.2-1);
- Unit 1 Division 12 ESF switchgear room (Fire Zone 5.1-1);
- Unit 2 Division A DG and day tank room (Fire Zone 9.2-2);
- Unit 2 Division B DG and day tank room (Fire Zone 9.2-1); and
- sprinkler head interference in 2B diesel oil storage tank room (Fire Zone 10.1-2).

The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Documents reviewed during this inspection are listed in the Attachment.

b. Findings

Failure to Maintain Fire Barrier in Accordance With Fire Protection Program

Introduction: The inspectors identified an Non-Cited Violation (NCV) of Braidwood Facility Operating License Nos. NPF-72 and NPF-77, Condition 2.E, for failing to maintain the firewall separating the fuel handling building and the auxiliary building, in accordance with the approved fire protection program.

Description: On April 12, 2006, during a routine fire protection walkdown of the fuel handling building, the inspectors noted that ventilation ducts in the 3-hour firewall between the spent fuel pool heat exchanger room and the auxiliary building did not

appear to have fire dampers installed. The inspectors questioned the fire protection system engineer regarding the state of the ventilation ducts in that area and their apparent lack of dampers. As a result, the licensee performed an independent walkdown and confirmed that fire dampers were not installed in the ventilation ducts of the firewall separating the spent fuel pool heat exchanger rooms of the fuel handling building and the Unit 1 and Unit 2 containment pipe penetration areas of the auxiliary building. This was inconsistent with Section 2.3.12.1 of the Fire Protection Report, which described the fire area analysis for the fuel handling building and stated that fire dampers were provided in the firewall separating the fuel handling building and the auxiliary building.

The inspectors reviewed the Fire Protection Report and did not identify any existing deviations allowing for the existence of this condition. The inspectors also reviewed Braidwood Station's Generic Letter 86-10 Evaluation, "Fire Protection Evaluation for Fire Zones 11.3-1/12.1-0 and 11.3-2/12.1-0 Boundaries to Demonstrate Separation Equivalent to Branch Technical Position CMEB 9.5-1, C5.b(2)," which was performed in response to the issue. The inspectors took into consideration the fact that the licensee's evaluation presented evidence in support of the ability of the existing condition to prevent the spread of fire from one zone to the other. Specifically referenced was ComEd Nuclear Design Information Transmittal MSD-97-021, dated December 17, 1997, which documented the ability of steel ventilation ducts and supports of a similar design to act as an effective fire barrier for a period of 60 minutes. The licensee entered the damper issue into their corrective action program for resolution, implemented compensatory measures that included hourly fire watches.

Analysis: The inspectors determined that the licensee's failure to maintain the firewall between the fuel handling building and the auxiliary building in accordance with the approved fire protection program was a performance deficiency warranting a significance determination. Furthermore, the issue was considered more than minor because the finding affected the attribute of protection against external factors (i.e. fire) of the Mitigating Systems Cornerstone. The inspectors assessed the finding using Inspection Manual Chapter 0609, Appendix F, Fire Protection Significance Determination Process, and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance because the steel ventilation ducts would provide a minimum of 60 minutes fire endurance protection, and the fixed fire ignition sources and combustibles were positioned such that the degraded barrier would not be subject to direct flame impingement.

Enforcement: Braidwood Station's Operating License Condition 2.E stated, in part, that "The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR." Section 9.5.1 of the UFSAR stated that "The design bases, system descriptions, safety evaluation, inspection and testing requirements, personnel qualification, and training are described in Reference 1 [the Fire Protection Report]." Section 2.3.12.1 of the Fire Protection Report stated, in part, that "Fire dampers are provided in the fire wall separating the fuel handling building and the auxiliary building." Contrary to the above, the licensee failed to have installed dampers in the firewall separating the spent fuel pool heat exchanger rooms of the fuel handling building and the Unit 1 and Unit 2 containment pipe penetration areas of the auxiliary building since original construction. Because this issue was entered into the

corrective action program as IR 477902, and the finding was of very low safety significance, this violation was being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000456/2006003-01; 05000457/2006003-01, Failure to Maintain Fire Barrier in Accordance with Fire Protection Program.

1R06 Flood Protection Measures (71111.06)

External Flooding Review

a. Inspection Scope

The inspectors reviewed Braidwood's flood analysis and design basis documents to identify design features important to external flood protection, and reviewed the external flood protection measures in place to prevent or mitigate effects of the probable maximum flood and the probable maximum precipitation. This included a general area walkdown of the outdoor plant area and perimeter to assess the condition and readiness of the various plant drainage system components to perform their function during a probable maximum flood or probable maximum precipitation scenario.

The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. This review represented one annual inspection sample. Documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

Annual Review

a. Inspection Scope

The inspectors reviewed the licensee's program for maintenance and testing of risk-important heat exchangers in the component cooling water system. Specifically, the review included the program for performance testing and analysis of the Unit 1 component cooling water heat exchanger when RH system shutdown cooling was established during the shutdown and subsequent cooldown to Mode 5. The inspectors observed the physical condition of the heat exchanger and performance testing apparatus and reviewed previous performance trend data to validate that the frequency of cleaning and testing was sufficient to detect degradation prior to loss of heat removal capabilities below design requirements; that the inspection results were appropriately categorized against pre-established engineering acceptance criteria, including the impact of tubes plugged on the heat exchanger performance; and that the licensee had developed adequate acceptance criteria for bio-fouling controls. This review represented one inspection sample. Additional documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From April 17, 2006, through May 5, 2006, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary, and the risk significant piping system boundaries for Unit 1. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the onsite inspection period.

For the following two types of nondestructive examination (NDE) activities the inspectors:

- observed ultrasonic test examination (UT) of the following welds to evaluate compliance with the ASME Code Section XI requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI:
  - feedwater welds (1FW-01-19, 20 and 33);
  - pressurizer shell-to-nozzle weld (N4A), nozzle inner radius, and upper shell welds (8E and 9D);
  - reactor coolant safe-end-to-elbow weld (1RC-32-1), pipe-to-elbow weld (1RC-32-3) and elbow-to-pipe weld (1RC-32-4); and
- reviewed dye penetrant examination report for a penetration-to-pipe weld (1SI-21-09) to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements.

The inspectors reviewed relevant indications (leakage) identified during a Code visual examination (VT)-2 from the previous outage at the excess letdown heat exchanger flanges to determine if the licensee's corrective actions and extent of condition reviews were in accordance with the ASME Code Section XI requirements.

The inspectors reviewed pressure boundary weld records for replacement of a 2 inch diameter safety injection system check valve (1SI-8819D) completed during the previous refueling outage, to determine if the welding acceptance and preservice examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements.

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Head Penetration (VHP) ISI

a. Inspection Scope

The inspectors did not perform a review of this procedure section (reduction in one inspection sample), because it is not required to be implemented until after completion of Temporary Instruction (TI) 2515/150, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles." Note that TI 2515/150 was implemented during this inspection.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From April 16, 2006, through April 27, 2006, the inspectors reviewed the Unit 1 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary.

The inspectors observed the licensee conducting a walkdown of borated systems within the Unit 1 containment outside the missile barrier. The scope of this walkdown included a bare metal visual examination of the reactor vessel closure head and vessel head penetrations from access doors on the service structure. The inspectors observed the licensee during these examinations to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action

requirements. In particular, the inspectors performed this observation to determine if the licensee focused BACC inspections on locations where boric acid leaks can cause degradation of safety significant components and to determine if degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed corrective actions and evaluations performed for boric acid found on reactor coolant system connected piping and components to confirm that corrective actions were consistent with requirements of Section XI of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI, and that the minimum Code required section thickness had been maintained for the affected components. In particular, this review focused on licensee corrective actions (reference IR 480489) implemented in response to identification of boric acid deposits on insulation and at four heater tube locations on the bottom of the Unit 1 pressurizer.

The documents reviewed during this inspection are listed in the Attachment to this report. The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

From April 21, 2006, through April 27, 2006, the inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements.

The NRC inspectors observed acquisition of eddy current test (ET) data, interviewed ET data analysts, observed in-situ pressure testing of degraded tubes and reviewed documents related to the SG ISI program to determine if:

- in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620, SG In-Situ Pressure Test Guidelines;
- in-situ pressure testing performance criterial were met for degraded tubes tested in SG A and SG B;
- the in-situ SG tube pressure testing screening criteria were properly applied in terms of SG tube selection based upon evaluation of the list of tubes with measured/sized flaws;
- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to identify tube degradation based on site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements and EPRI 1003138, Pressurized Water Reactor SG Examination Guidelines, Revision 6;

- the SG tube ET examination scope included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, and support plates;
- the licensee identified new tube degradation mechanisms;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements;
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;
- the licensee did an evaluation for unretrievable loose parts;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, Performance Demonstration for Eddy Current Examination, of EPRI 1003138, Pressurized Water Reactor SG Examination Guidelines, Revision 6; and
- the licensee identified deviations from ET data acquisition or analysis procedures.

The inspectors performed a review of SG ISI related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the SG related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to SG tube integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, Corrective Action, requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Quarterly Review of Testing/Training Activity

a. Inspection Scope

The inspectors observed operating crew performance during an evaluated simulator out-of-the-box scenario involving multiple solid state protection system input bistable failures requiring plant shutdown, with subsequent fuel failure during unit ramp down.

The inspectors evaluated crew performance in the following areas:

- clarity and formality of communications;
- ability to take timely actions in the safe direction;
- prioritization, interpretation, and verification of alarms;
- procedure use;
- control board manipulations;
- oversight and direction from supervisors; and
- group dynamics.

Crew performance in these areas was compared to licensee management expectations and guidelines.

The inspectors verified that the crew completed the critical tasks listed in the simulator guide. The inspectors also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors observed the licensee evaluators to determine whether they also noted the issues and discussed them in the critique at the end of the session. Documents reviewed are listed in the Attachment. This review constituted one sample of this inspection requirement.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Inspection

a. Inspection Scope

The inspectors reviewed the licensee's overall maintenance effectiveness for selected plant systems. This evaluation consisted of the following specific activities:

- observing the conduct of planned and emergent maintenance activities where possible;
- reviewing selected IRs, open work orders, and control room log entries in order to identify system deficiencies;
- reviewing licensee system monitoring and trend reports;
- attending various meetings throughout the inspection period where the status of maintenance rule activities was discussed;
- a partial walkdown of the selected system; and
- interviews with the appropriate system engineer.

The inspectors also reviewed whether the licensee properly implemented Maintenance Rule, 10 CFR 50.65, for the chosen systems. Specifically, the inspectors determined whether:

- the system was scoped in accordance with 10 CFR 50.65;
- performance problems constituted maintenance rule functional failures;

- the system had been assigned the proper safety significance classification;
- the system was properly classified as (a)(1) or (a)(2); and
- the goals and corrective actions for the system were appropriate.

The above aspects were evaluated using the maintenance rule program and other documents listed in the Attachment. The inspectors also verified that the licensee was appropriately tracking reliability and/or unavailability for the systems.

The inspectors completed two samples in this inspection requirement by reviewing the following systems:

- 0A and 0B control room ventilation trains subsequent to a fuse failure resulting in a maintenance rule functional failure; and
- Unit 1 and Unit 2 main power systems subsequent to an increasing trend in licensee identified main power transformer deficiencies.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's management of plant risk during emergent maintenance activities or during activities where more than one significant system or train was unavailable. The activities were chosen based on their potential impact on increasing the probability of an initiating event or impacting the operation of safety-significant equipment. The inspections were conducted to determine whether evaluation, planning, control, and performance of the work were done in a manner to reduce the risk and minimize the duration where practical, and that contingency plans were in place where appropriate.

The licensee's daily configuration risk assessment records, observations of operator turnover and plan-of-the-day meetings, and observations of work in progress, were used by the inspectors to verify that; the equipment configurations were properly listed; protected equipment were identified and were being controlled where appropriate; work was being conducted properly; and significant aspects of plant risk were being communicated to the necessary personnel. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

In addition, the inspectors reviewed selected issues, listed in the Attachment, that the licensee encountered during the activities, to determine whether problems were being entered into the corrective action program with the appropriate characterization and significance.

The inspectors completed six samples by reviewing the following activities:

- delayed return to service of 1B DG following pushrod replacement;
- review of licensee risk assessment for transition to Mode 4 with the 1A essential service water pump inoperable;
- 1B RH train outage due to overhaul of recirculation sump isolation valve operator 1SI8811B;
- Unit 0 component cooling water heat exchanger outage, which results in dual unit yellow risk condition;
- 2B DG outage for periodic engine and generator overhaul; and
- 2A CS pump mechanical seal replacement following gross leakage during initial post maintenance testing.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors completed one sample by observing and/or reviewing operator performance during the Unit 2 25B feedwater heater drain cooler relief valve failure on April 6, 2006.

The inspectors observed the control room response, interviewed plant operators and reviewed plant records including control room logs, operator turnovers, and IRs. The inspectors verified that the control room operators' response was consistent with station procedures and that identified discrepancies were captured in the corrective action program.

The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program. Documents reviewed as part of this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated plant conditions and selected IRs for risk-significant components and systems in which operability issues were questioned. These conditions were evaluated to determine whether the operability of components was justified. The inspectors compared the operability and design criteria in the appropriate section of the UFSAR to the licensee's evaluations presented in the IRs and documents listed in the Attachment to verify that the components or systems were operable. The

inspectors also conducted interviews with the appropriate licensee system engineers and conducted plant walkdowns, as necessary, to obtain further information regarding operability questions. Documents reviewed as part of this inspection are listed in the Attachment.

The inspectors completed three samples by reviewing the following operability evaluations and conditions:

- repeated cracking of reactor containment fan cooler (RCFC) turning vanes;
- scaffolding adjacent to 1B DG essential service water piping; and
- elevated spent fuel pool temperature conditions.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance testing activities associated with important mitigating systems, barrier integrity, and support systems to ensure that the testing adequately demonstrated system operability and functional capability. The inspectors used the appropriate sections of the TS and UFSAR, as well as the Work Orders for the work performed, to evaluate the scope of the maintenance and to determine whether the post-maintenance testing was performed adequately, demonstrated that the maintenance was successful, and that operability was restored. The inspectors determined whether the tests were conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the tests were properly reviewed and recorded. The activities were selected based on their importance in demonstrating mitigating systems capability and barrier integrity. Documents reviewed as part of this inspection are listed in the Attachment.

Six samples were completed by observing post-maintenance testing of the following components:

- 1B DG start-up subsequent to engine pushrod replacement;
- 1SI8811B valve automatic actuation testing subsequent to motor operator maintenance;
- 1MS018D 1D SG power operated relief valve stroke testing subsequent to valve work outage;
- Unit 0 component cooling water heat exchanger leak check and operation subsequent to piping replacement of tube side vent lines;
- 2A component cooling water pump ASME run subsequent to pump motor replacement and re-balance; and
- 2B DG fast start and engine power factor testing subsequent to 6 year engine and generator overhaul.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan and contingency plans for the Unit 1 refueling outage, conducted April 16 - May 3, 2006, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Documents reviewed during the inspection are listed in the Attachment. This inspection constituted one sample.

This inspection included:

- initial walkdown of containment to look for evidence of reactor coolant system leakage and other discrepancies;
- review of licensee configuration management, including maintenance of defense-in-depth commensurate with the Outage Safety Plan for key safety functions and compliance with the applicable TS when taking equipment out of service;
- observation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- review of the installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error;
- review of the licensee's controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes;
- review of the licensee's controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- monitoring reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- monitoring the licensee's controls over activities that could affect reactivity;
- observations of maintenance on secondary containment as required by TS;
- observation and review of refueling activities, including fuel handling;
- observation and review of startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- monitoring and review of licensee identification and resolution of problems related to refueling outage activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed surveillance testing activities associated with important mitigating systems, barrier integrity, and support systems to ensure that the testing adequately demonstrated system operability and functional capability. The inspectors used the appropriate sections of the TS and UFSAR to determine whether the surveillance testing was performed adequately and that operability was restored. The inspectors determined whether the testing met the frequency requirements; that the tests were conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the tests were properly reviewed and recorded. The activities were selected based on their importance in demonstrating mitigating systems capability, barrier integrity and the initiating events cornerstone. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Documents reviewed as part of this inspection are listed in the Attachment.

Six samples were completed by observing and evaluating the following surveillance tests:

- OB fire protection pump flow and pressure test;
- Unit 1 main steam safety valve lift testing (inservice testing sample);
- 1B DG emergency core cooling system sequencer testing;
- Unit 1 control rod drop time testing;
- 2B DG slave relay 611B fast start and engine overspeed testing; and
- Unit 2 reactor coolant system leak rate calculation (reactor coolant leakage detection sample).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the installation of temporary storage tanks for liquid radwaste processing.

For the above modification, the inspectors reviewed the associated design change paperwork, performed a walkdown of the tanks and associated piping; observed the transfer of liquid radwaste to the tanks and discussed radiological and environmental controls with applicable engineering, operations, and radiation protection staff. The inspectors also verified that minor issues identified during the inspection were entered

into the licensee's corrective action program. Those documents reviewed during this inspection are listed in the Attachment. This review constituted one sample of this inspection requirement.

b. Findings

No findings of significance were identified, however, a minor violation associated with this temporary modification was described in Section 4OA2.2 of this report.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed licensee performance during one crew emergency preparedness drill on the simulator and one site emergency preparedness drill on the simulator and Technical Support Center for a total of two samples. The inspectors observed event classification, notification, and development of protective action recommendations, manning of the emergency response facilities, and turnover of command and control. The inspectors also observed portions of the post drill critiques to determine whether their observations were also identified by the licensee evaluators and reviewed documents listed in the Attachment to determine whether deficiencies were entered into the licensee's corrective action system.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors discussed performance indicators with the radiation protection staff and reviewed data from the licensee's corrective action program to determine if there were any performance indicator occurrences in the occupational exposure cornerstone that had not been reported or reviewed. The inspector limited this review to incidents occurring since the last inspection in this area, (June 2005). The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program. Documents reviewed during this inspection are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns, Observation of Radiological Access Controls and Radiation Work Permits

a. Inspection Scope

The inspectors selected several radiologically significant activities for further review. These activities included those having significant total exposure estimates and/or being performed in high radiation or potential airborne areas in the plant. Selected work packages and radiation work permits (RWPs) were reviewed to determine if appropriate controls (i.e., surveys, postings and barricades) were being used. This review represented one sample.

The inspectors performed a walkdown of radiological controlled areas in the auxiliary building and Unit 1 containment, to observe whether licensee surveys were complete and accurate and whether radiological areas were properly posted. For selected activities, the inspectors reviewed RWPs, and observed ongoing work to determine if areas of significant radiological concern, (such as hot spots or higher dose rate areas) were properly identified. Several workers were also interviewed to verify that they understood the RWP requirements and radiological conditions in their work area. This review represented one sample.

The inspectors performed a walkdown of selected high radiation areas and all areas currently being controlled as a locked or very high radiation area. This walkdown included the auxiliary building and Unit 1 containment, but not the Unit 2 containment. Specifically, the inspectors observed whether postings and barriers were properly used to control access to these areas. The inspectors also selectively observed whether electronic dosimetry was properly used by workers in these areas and, through interviews, whether the workers were aware of the dosimetry alarm setpoints and access control requirements. Site TSs and the following station procedures were used as standards for the appropriate barriers and controls:

- RP-AA-460, Controls for High and Very High Radiation Areas, Revision 10;
- RP-AA-376, Radiological Postings, Labeling, and Markings, Revision 1; and
- RP-AA-376-1001, Radiological Posting, Labeling, and Marking Standard, Revision 3.

This review represented one sample.

The inspectors reviewed the following activities and evaluated the radiological controls to determine whether workers were adequately protected against airborne contamination:

- reactor cavity decontamination;
- under pressurizer weld examination and No.15 heater removal; and
- SG work (manway and diaphragm removal/installation).

These activities were selected as they had the potential for workers to receive an internal exposure of greater than 50 millirem committed effective dose equivalent. The inspectors reviewed the associated as-low-as-is-reasonably-achievable (ALARA) plans and RWPs, and observed work activities to evaluate whether engineering controls, (such as high efficiency particulate air filtration and respirators) were appropriately considered and used. Field observations were also used to verify that air samplers were appropriately placed and operational. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a recently completed licensee self-assessment that focused on high radiation area controls and reviewed condition reports generated since the last inspection (June 2005), related to access control or high radiation area incidents, to determine if identified problems were being entered into the licensee's corrective action program. This review represented one sample.

Issue reports related to access controls or high radiation area incidents were reviewed to determine if they were being properly evaluated. Specifically, the issue reports were reviewed against the following criteria:

- initial problem identification and screening;
- disposition of potential operability/reportability issues;
- evaluation of safety significance and/or risk;
- identification of cause; and
- implementation of corrective actions.

This review also considered whether recurring events or adverse trends were properly evaluated and addressed. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance and Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations the inspectors evaluated radiation worker performance with respect to stated RWP work requirements. Specifically, whether workers were aware of the radiological hazards present and whether they were properly

utilizing those controls implemented to protect against such hazards. This review represented one sample.

During walkdowns of the auxiliary building and Unit 1 containment, the inspectors evaluated radiation protection technician performance. Specifically, the inspectors determined if technicians adequately covered work activities, performed radiological surveys and briefed workers on radiological conditions. The inspectors also interviewed several technicians to verify that RWP requirements were well understood. This review represented one sample.

Documents reviewed as part of this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable Planning and Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors reviewed the following Unit 1 work activities to determine the efficacy of the licensee's ALARA planning:

- SG work;
- reactor head disassembly/reassembly;
- control rod drive mechanism volumetric inspections;
- snubber removal, inspection and testing;
- scaffold staging, building and removal;
- pressurizer boric acid inspection and cleaning; and
- replacement of the No. 15 pressurizer heater.

These activities were selected based on their estimated total exposure, potential for significant radiological conditions (airborne, work in high radiation areas, etc.) and potential for emergent activities. The inspectors used the guidance contained in licensee procedures RP-AA-400, ALARA Program, Revision 3 and RP-AA-401, Operational ALARA Planning and Controls, Revision 5, as the criteria for the review.

The inspectors evaluated whether the RWPs were consistent with the associated ALARA plans for the above activities. In particular, whether electronic dosimeter dose and dose rate alarm setpoints were appropriate given the expected work area radiological conditions. Additionally, the inspectors noted whether engineering controls credited in the ALARA plan were appropriately captured in the RWP. The inspectors also observed whether the RWP and ALARA plan requirements were properly communicated during pre-job briefings. This review represented one sample.

The inspectors compared the actual dose received against the estimated dose for the above ALARA plans. These comparisons were made to gauge the accuracy of the

licensee's dose estimates based on knowledge of the work activity. Specifically, whether the licensee properly used previous work history and/or information from other work groups (such as man-hour estimates) in dose estimates. The inspectors also reviewed work-in-progress and post job reviews. Reasons for inconsistencies between the actual and intended dose were discussed with radiation protection staff to determine whether the differences resulted from radiation controls or job planning. This review represented one sample.

The inspectors evaluated the licensee's dose reduction strategies utilized in the above ALARA plans. In particular, whether shielding from water filled components and piping, job scheduling and coordination with shielding and scaffold installation and removal were considered. Temporary shielding requests were also evaluated with respect to dose rate reduction, along with engineering shielding responses follow-up. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Job Site Inspection and ALARA controls

a. Inspection Scope

Electronic dose reports from workers involved in the activities reviewed under Section .1 above, were reviewed to determine if there were any significant exposure variations. Specifically, these variations were evaluated to determine if they were caused by poor ALARA work practices or by differences in job skill assignments. These evaluations also consisted of observing selected work activities to monitor worker performance. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed the source term reduction actions implemented by the licensee for the Unit 1 refueling outage. These actions included hydrogen peroxide addition, reactor coolant filtration and hydrolazing. The effectiveness of these actions were, in part, evaluated by comparing the observed average plant dose rates from the current outage to historical trends. The effectiveness of these actions on those work activities observed by the inspectors were also considered. The plant source term (including input mechanisms) and overall mitigation strategies were discussed with radiation protection staff. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Declared Pregnant Workers

a. Inspection Scope

The inspectors reviewed the licensee's program for monitoring the exposure of declared pregnant workers. This program was described in RP-AA-270, Prenatal Radiation Exposure, Revision 3. The inspectors determined whether the program was consistent with the requirements of 10 CFR 20.1208. There were no declared pregnant workers during this assessment period. Documents reviewed are listed in the Attachment. This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the following licensee self-assessments:

- source term reduction, dated June 2004; and
- SG outage ALARA report for A2R11, dated Spring 2005

The inspectors also reviewed condition reports generated since the last inspection (June 2005), related to ALARA planning or source term reduction, to determine if identified problems were being entered into the licensee's corrective action program. This review represented one sample.

The inspectors determined if identified problems were being entered into the corrective action program for resolution. This included dose significant work-in-progress and post-job reviews of exposure performance. This review represented one sample.

Issue reports related to the ALARA program were reviewed to determine if they were being properly evaluated. Specifically, the issue reports were reviewed against the following criteria:

- initial problem identification and screening;
- disposition of potential operability/reportability issues;
- evaluation of safety significance and/or risk;
- identification of cause; and
- implementation of corrective actions.

This review also considered whether recurring events or adverse trends were properly evaluated and addressed. This review represented one sample.

Documents reviewed as part of this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

Tritiated Liquid Discharge Storage, Monitoring, and Remediation

a. Inspection Scope

As discussed previously in Inspection Reports 05000456/2005010; 05000457/2005010; Section 4OA3.1, and 05000456/2006002; 05000457/2006002; Section 4OA3.1, the inspectors continued to monitor the licensee activities resulting from previous inadvertent leaks of tritiated liquid from the blowdown line to the Kankakee River. This inspection was not considered a complete sample. The inspection activities included completion or review of the following:

Temporary Storage of Liquid

- results of licensee fixed rear axil container (FRAC) tank inspections;
- safety evaluation of installation of FRAC tanks; and
- several walkdowns of FRAC tank installation and transfer hoses.

Mitigation of Previous Spills

- disposal of water from vacuum breaker vaults and off-site cistern;
- installation of concrete bottoms and waterproof membranes in vacuum breaker vaults;
- installation of isolation/throttle valve at river end of blowdown line;
- tie-in of pond pumping pipe to blowdown line in vacuum breaker #2 vault;
- installation and testing of vacuum breaker vault leakage alarms;
- installation and testing of pond pump;
- installation and testing of the pond pumping composite sampler;
- implementation of state injunction order;
- revisions to the Offsite Dose Calculation Manual for pond pumping;
- response to vacuum breaker water alarms;
- startup and operation of pond pumping system;
- procedures for obtaining composite samples of pumped pond water; and
- several inspections of vacuum breaker vaults during pumping operations.

Response to New Spills

- 25B drain cooler relief valve lift on April 6, 2006;
- small spill from auxiliary boiler test line on May 15, 2006;
- procedures for monitoring waste water discharges to the cooling lake; and
- procedures for spill notification.

In addition, the inspectors attended and presented information at several meetings, hosted by the licensee, for interested community members, and accompanied public officials on tours of the affected areas. The inspectors also obtained numerous split water samples from the licensee and sent them to an independent laboratory for analysis. The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program. Documents reviewed as part of this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified, however, a minor violation associated with temporary modification that installed the FRAC tanks was described in Section 4OA2.2 of this report.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors reviewed the document listed in the Attachment to verify that the licensee had correctly reported Performance Indicator data, in accordance with the criteria in Nuclear Energy Institute 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2. The data reported by the licensee was compared to a sampling of control room logs, IRs, Licensee Event Reports, and other sources of data generated since the last verification. The inspectors completed two samples by reviewing the following Performance Indicators:

- Unit 1 safety system functional failures from July 1, 2004, through March 31, 2006; and
- Unit 2 safety system functional failures from July 1, 2004, through March 31, 2006.

Cornerstone: Barrier Integrity

The inspectors reviewed the documents listed in the Attachment to verify that the licensee had correctly reported Performance Indicator data, in accordance with the criteria in Nuclear Energy Institute 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2. The data reported by the licensee was compared to a sampling of control room logs, chemistry records, surveillance records, and other sources of data generated since the last verification. The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program. The inspectors completed four samples by reviewing the following Performance Indicators:

- Unit 1 reactor coolant system specific activity from July 1, 2004, through March 31, 2006;

- Unit 2 reactor coolant system specific activity from July 1, 2004, through March 31, 2006;
- Unit 1 reactor coolant system leak rate from April 1, 2004, through March 31, 2006; and
- Unit 2 reactor coolant system leak rate from April 1, 2004, through March 31, 2006.

b. Findings

No findings of significance were identified.

Note that this inspection covered data reported for about 2 years since the inspection requirement was waived in 2005.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to determine whether they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action program as a result of the inspectors' observations are generally denoted in the Attachment. These activities were part of normal inspection activities and were not considered separate samples.

b. Findings

No findings of significance were identified.

.2 Annual Sample - Issues Related to Temporary Storage of Liquid Radioactive Waste

Introduction

The inspectors reviewed the adequacy of the licensee's corrective actions, prioritization, and evaluation of issues related to the onsite, temporary storage of liquid radioactive waste. The liquid was being stored in temporary tanks commonly referred to as FRAC tanks. Temporary storage was required following the suspension of routine liquid radioactive waste discharges after discovery of leaks from the circulating water blow down line. The issues associated with the blowdown line leaks, including the discovery, licensee actions, and the results of an NRC inspection are documented in NRC Inspection Reports 05000456/2005010, 05000457/2005010, 05000456/2006002, 05000457/2006002, 05000456/2006008(DRS) and 05000457/2006008(DRS). The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. Those documents reviewed during this inspection are listed in the Attachment. This activity completed one sample.

Observations

The inspectors assessed the licensee's evaluation, including the review performed in accordance with 10 CFR Part 50.59, Changes, Tests and Experiments. The inspectors determined that the licensee's conclusion, "This activity does not need NRC approval prior to implementation," was appropriate. The inspectors' assessment also determined that the installation of the FRAC tanks did not comply with Regulatory Guide 1.143, Revision 0, however the licensee had already reached that conclusion prior to the onsite portion of this inspection.

On April 13, 2006, the licensee's Nuclear Oversight group determined that Revision 0 of the 10 CFR 50.59 review failed to address Regulatory Guide 1.143. The observation was documented in the licensee's corrective action system and subsequent revisions to the 10 CFR 50.59 review did address Regulatory Guide 1.43. The original 10 CFR 50.59 review failed to recognize that the commitment to follow Regulatory Guide 1.143, Revision 0, was explicitly described in the UFSAR and therefore the guidance contained in the regulatory guide needed to be addressed. Based on the inspectors' review, it was determined that the initial 10 CFR 50.59 review was inadequate since it failed to address the commitment to Regulatory Guide 1.143. The safety significance of the noncompliance with Regulatory Guide 1.143 was minor because the licensee's revision to the 10 CFR 50.59 that addressed Regulatory Guide 1.143 concluded that NRC approval was still not required. Specifically, the licensee's revision to the 10 CFR 50.59 review addressed the aspects of the regulatory guide that were not being met. The new 10 CFR 50.59 provided sufficient basis or compensatory actions to conclude that the installation of the temporary storage tanks did not require prior NRC approval in accordance with 10 CFR 50.59. The inspectors reviewed the revised 10 CFR 50.59 evaluation and acknowledged that the licensee appropriately determined that prior NRC approval was not required. Therefore, this issue constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the Enforcement Policy.

.3 Semiannual Review to Identify Trends

a. Inspection Scope

The inspectors performed a review of the licensee's Corrective Action Program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. An issue report trend review focused on systems important to risk according to the licensee's probabilistic risk assessment model was performed for main power, component cooling water, instrument/service air, circulating water, switchyard, essential service water, reactor cooling, auxiliary feedwater, pressurizer, and safety injection systems. The review also included issues documented outside the normal CAP including focus area self assessments, corrective maintenance backlog reports, common cause analysis reports, component status reports, and maintenance rule assessments. The inspectors' review nominally considered the 6-month period of January through June 2006, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors

compared and contrasted their results with the results contained in the licensee's mechanisms for identifying and correcting trends. Corrective actions associated with a sample of the issues identified by the licensee were also reviewed for adequacy. Specific documents reviewed are listed in the Attachment.

b. Assessment and Observations

Overall the inspectors' review noted that the licensee aggressively identified trends through diverse means. System level trends were routinely identified by system engineering through the site's CAP. Programmatic trends were generally identified in a timely manner by nuclear oversight, operations, or engineering through the focus area self assessment process or via the CAP. The inspectors' daily issue report review along with the documents reviewed specifically for this sample did not indicate the existence of a trend not previously identified by the licensee.

4OA5 Other Activities

.1 Reactor Pressure Vessel Head and VHP Nozzles (TI 2515/150)

a. Inspection Scope

On February 11, 2003, the NRC issued Order EA-03-009 (ADAMS Accession Number ML030410402). This order required examination of the reactor vessel closure head (RVCH) and associated VHP nozzles to detect primary water stress corrosion cracking (PWSCC) of VHP nozzles and corrosion of the vessel head. The purpose of TI 2515/150, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles," was to implement an NRC review of the licensee's head and VHP nozzle inspection activities required by NRC Order EA-03-009.

The inspectors performed a review in accordance with TI 2515/150 of the licensee's procedures, equipment, and personnel used for examinations of the RVCH and VHP nozzles to confirm that the licensee met requirements of NRC Order EA-03-009 (as revised by NRC letter dated February 20, 2004). The results of the inspectors' review included documentation of observations in response to the questions identified in TI 2515/150.

From April 19, 2006, through April 24, 2006, the inspectors performed a review of the licensee's RVCH inspection activities completed in response to NRC Order EA-03-009. This review included:

- observation of the licensee personnel conducting automated UT and ET of VHP nozzle locations from the on-site data acquisition trailer;
- interviews with NDE personnel performing examinations of the RVCH and VHP nozzles from an on-site trailer;
- certification records of NDE personnel performing examinations of the RVCH and VHP nozzles;
- UT and ET examination procedures used for examinations of the RVCH and VHP nozzles;

- procedures used for identification and resolution of boric acid leakage from systems and components above the vessel head;
- the licensee's procedures and corrective actions for boric acid leakage; and
- UT and ET examination records for the RVCH and VHP nozzles.

The inspectors conducted these reviews to confirm that the licensee performed the vessel head examinations in accordance with requirements of NRC Order EA-03-009, using procedures, equipment, and personnel qualified for the detection of PWSCC in vessel VHP nozzles and detection of vessel head wastage.

From April 17, 2006, through April 26, 2006, the inspectors reviewed the licensee's VHP nozzle susceptibility ranking calculation to:

- verify that appropriate plant-specific information was used as input;
- confirm the basis for the head temperature used by licensee; and
- determine if previous VHP cracks had been identified, and if so, documented in the susceptibility ranking calculation.

The documents reviewed by the inspectors in conducting this inspection are listed in the Attachment to this report.

b. Observations

Summary: At of the end of operating cycle No. 12, the Braidwood Unit 1 vessel head was at 2.2 effective degradation years (EDY), which is in the low susceptibility ranking category as described in NRC Order EA-03-009. To meet the inspection requirements of Order EA-03-009, the licensee completed automated UT and ET examinations for each of the 78 VHP nozzles and head vent line. The licensee identified nine vessel head penetrations with minor limitations in the volumetric examination coverage below the J-groove weld required by Order EA-03-009. Additionally, at the inside surface of VHP nozzle No. 74 a surface anomaly was identified, which limited examination coverage in an area above the J-groove weld. The inspectors were also concerned that the disrupted metal area at the inside surface of VHP nozzle No. 74 may require further evaluations to determine if it could facilitate the onset of PWSCC. Following restart of Unit 1, the licensee intended to request relaxation from the Order to accept the VHP nozzles with limited examinations.

Overall, the inspectors concluded that the licensee had completed an examination of the reactor vessel head using methods which were consistent with the requirements of Order EA-03-009. The inspectors' responses and conclusions to specific questions identified in TI-2515/150 related to the quality of personnel, procedures, and equipment used to perform the vessel head examination are discussed below. The inspectors could not independently confirm the ability of some of the NDE techniques to detect PWSCC. This condition reflected a lack of industry or vendor "qualified" techniques and did not represent a deviation from NRC Order EA-03-009, which did not specify qualification or demonstration standards for the NDE techniques used. Additionally, the inability to identify PWSCC within the J-groove weld is consistent with the requirements of Order EA-03-009, which does not require examination of the J-groove welds when UT of the nozzle base material has been completed.

Evaluation of Inspection Requirements

In accordance with the reporting requirements contained within TI 2515/150, Revision 3, the inspectors evaluated and answered the following questions:

a. For each of the examination methods used during the outage, was the examination:

1. Performed by qualified and knowledgeable personnel?

Yes. The licensee's vendor NDE staff that performed the automated UT and ET examinations were certified to a level II or level III for these examinations. The licensee vendor certified their NDE staff in accordance with vendor Procedures WDP-9.2, Qualification and Certification of Personnel in Nondestructive Examination and accepted subcontracted NDE staff qualified to different recommended practices which met industry standard ANSI/ASNT CP-189-1991 ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel.

2. Performed in accordance with demonstrated procedures?

Yes. The licensee's vendor performed automated UT and ET of VHP nozzles in accordance with Procedure WDI-UT-010, Intraspect Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic, Longitudinal Wave, and Shear Wave, Revision 12. The vendor performed these examinations from the inside nozzle surface using probes which contained UT and ET equipment configurations which were consistent with those used during vendor mockup testing. This procedure identified a number of UT probes which could be used for this examination, but it did not identify the specific probes or equipment settings which had been demonstrated. For the Braidwood Unit 1 vessel head examination, the inspectors verified that the vendor used UT probes, frequencies, and angles that were consistent with that used in the demonstration.

The licensee's vendor had demonstrated an earlier version of procedure WDI-UT-010 on mockup VHP nozzles which contained cracks or simulated cracks as documented in EPRI MRP-89, Materials Reliability Program Demonstrations of Vendor Equipment and Procedures for the Inspection of Control Rod Drive Mechanism Head Penetrations. The inspectors reviewed the summary of changes up through Revision 12 of Procedure WDI-UT-010 from Revision 3, which had been demonstrated as documented in EPRI MRP-89, to ensure that any equipment configuration changes did not affect flaw detection capability. Additionally, the inspector reviewed the vendor's technical justifications for changes in equipment configurations (e.g., changes in cable length, or eddy current probe frequencies) that could affect

detection capability. These supporting vendor technical documents reviewed during this inspection are listed in the Attachment to this report.

3. Able to identify, disposition, and resolve deficiencies and capable of identifying the PWSCC and/or head corrosion phenomena described in Order EA-03-009?

Automated UT/ET of VHP Nozzles Equipped with a Thermal Sleeve

Yes. The licensee's vendor examined the 55 sleeved control rod drive VHP nozzle base metal using a Trinity Blade Probe from the inside surface of the nozzles. The Trinity Blade Probe contained a time-of-flight-diffraction UT transducer, a zero degree UT transducer, and an ET coil designed to optimize detection of both circumferential and axial oriented flaws. The UT portion of this probe was also configured to detect leakage paths in the shrink fit region between the VHP nozzle tube and the reactor vessel head material. The licensee's vendor had detected PWSCC in VHP nozzles at Beaver Valley Unit 1 as documented in PVP2004-2555, Advanced Nondestructive Examination Technologies for Alloy 600 Components, using this examination technique. The licensee had also detected simulated flaws in VHP mockups as documented in EPRI MRP-89 using this technique. Therefore, the inspectors concluded that this examination would have been effective for detection of PWSCC in the Braidwood Unit 1 VHPs.

Automated UT/ET of VHP Nozzles without a Thermal Sleeve

Yes. The licensee's vendor examined the 23 unsleeved control rod drive VHP nozzle base metal using a rotating probe from the inside surface. This probe contained time-of-flight-diffraction UT transducer pairs, zero degree UT transducers, and ET coils designed to optimize detection of both circumferential and axial oriented flaws. The UT portion of this probe was also configured to detect leakage paths in the shrink fit region between the VHP nozzle tube and the reactor vessel head material. The licensee's vendor had detected PWSCC in VHP nozzles at Beaver Valley Unit 1 as documented in PVP2004-2555, Advanced Nondestructive Examination Technologies for Alloy 600 Components, using this examination technique. The licensee had also detected simulated flaws in VHP mockups as documented in EPRI MRP-89 using this technique. Therefore, the inspectors concluded that this examination would have been effective for detection of PWSCC in the Braidwood Unit 1 VHPs.

Vent Line Penetration ET

Unknown. The licensee's vendor used probes containing an array of ET coils to examine the inside of the head vent line and vent line VHP nozzle J-groove weld. This technique had been used on a vendor mockup and on a calibration standard which both contained electric discharge machined notches. Because this demonstration did not include actual or closely simulated PWSCC type flaws, the inspectors could not independently confirm that this examination would have been effective at detection of PWSCC. Additionally, this equipment was not equipped with ET probes which could detect outside diameter initiated circumferentially oriented cracking.

VHP Nozzle J-Groove Welds

No. The licensee's vendor examinations of the VHP nozzle base material were not designed to detect PWSCC contained entirely within the VHP nozzle J-groove welds. Based upon a review of vendor equipment performance capability on simulated cracks documented in EPRI MRP-89, the UT techniques generally could not consistently detect cracking until it had reached 10 percent or greater depth into the VHP nozzle thickness. Therefore, the inspectors concluded that these examinations would not be effective at identification of PWSCC flaws located entirely within the J-groove weld. However, the licensee did implement a demonstrated UT technique intended to identify evidence of leakage behind a VHP nozzle caused by through-wall cracking of the J-groove weld.

b. What was the physical condition of the reactor vessel head (e.g., debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The licensee was not required by the NRC Order EA-03-009 to conduct a qualified visual examination of the Braidwood Unit 1 vessel head during this refueling outage. Although not required by the Order, the licensee performed an inspection of the bare metal head to meet the station's boric acid program and VHP nozzles through the access doors in the service structure. Based upon this inspection, the licensee did not identify any indication of boric acid leakage from sources above the vessel head. The inspectors observed the head during this inspection and did not observe any evidence of boric acid leakage. The inspectors noted some areas of minor staining on the VHP nozzles which the licensee had noted during prior inspections.

c. Could small boron deposits, as described in the Bulletin 01-01, be identified and characterized?

Not applicable. The licensee performed a volumetric examination of the reactor from under the vessel head during the refueling outage and did not perform a qualified bare metal visual examination as discussed above.

d. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

None.

e. What, if any, impediments to effective examinations, for each of the applied methods, were identified (e.g., centering rings, insulation, thermal sleeves, instrumentation, nozzle distortion)?

The licensee identified physical limitations (due to RVCH and VHP nozzle design configurations) to completing the extent of the examination coverage required by NRC Order EA-03-009. Specifically, the licensee could not meet the NRC Order EA-03-009, requirement IV.C.(5)(i) to perform ultrasonic testing to at least 1 inch below the toe of the J-groove weld for 9 VHP nozzles. The extent of coverage achieved below the toe of the J-groove weld for these VHP nozzles was less than 1 inch due to the short distance that these nozzles extended below the J-groove welds and/or the presence of threads on the outside surface of these nozzles. Because these nonvisual examinations were completed earlier than required under the NRC Order EA-03-009, the licensee did not need to rely on the inspection results to remain in compliance with the NRC Order prior to restart. To remain in compliance with the NRC Order, the licensee intended to request relaxation from the NRC Order EA-03-009 requirements for these VHP nozzles with limitations after restart and before the next refueling outage.

The licensee also identified an area above the J-groove weld in VHP nozzle No. 74, which could not be examined with UT or ET probes due to an irregular surface condition. The licensee conducted a video probe assisted visual examination, which revealed scoring and metal disruption at the inside surface of this penetration. Based upon ET data the inspectors estimated that an area of disrupted material existed that was approximately 10 degrees (0.25 inches) in circumferential extent and 0.6 inches in height. The licensee believed that this area was caused by galling of the inside surface when a spring clip became wedged against the thermocouple housing during maintenance activities which occurred at least 12 years ago. The inspectors were concerned that the metal disruption could serve to make this area of VHP nozzle No. 74 more susceptible to PWSCC. The licensee intended to address the lack of coverage for this area during a relaxation request from the NRC Order EA 03-009 and intended to document the nozzle limitations and the surface condition of VHP nozzle No. 74 in their corrective action system.

f. What was the basis for the temperatures used in the susceptibility ranking calculation, were they plant-specific measurements, generic calculations, (e.g., thermal hydraulic modeling, instrument uncertainties), etc.?

NRC Order EA-03-009 required licensees to calculate the susceptibility category of the reactor head to PWSCC-related degradation. The susceptibility category in EDY establishes the basis for the vessel head examination schedule and scope. In May of 2005, the licensee calculated the EDY for the Braidwood Unit 1 reactor head as documented in work order 0070306. In this calculation, the

licensee used the formula required by NRC Order EA-03-009 and determined the EDY for the vessel head for several operating cycles. Based upon this calculation, at the end of operating cycle No.12, the Braidwood Unit 1 reactor vessel head was predicted to reach 2.2 EDY, which placed it in the low susceptibility category.

The NRC Order EA 03-009 Section IV.A required "This calculation shall be performed with best estimate values for each parameter at the end of each operating cycle for the RVCH that will be inservice during the subsequent operating cycle." Contrary to this requirement, as documented in Work Order 0070306, the licensee used estimated data for reactor power level and cycle length approximately 11 months prior to the end of operating cycle No. 12. The licensee entered this issue into the corrective action system (AR 00483826) and re-performed this calculation on April 29, 2006. This issue was considered a violation of NRC Order EA-03-009 of minor significance, because the revised calculation did not affect the original 2.2 EDY estimate and hence did not affect the head examination requirements.

NRC Order EA-03-009, required the licensee to have used best estimate values for the vessel head temperature in the EDY calculation. From the design average reactor coolant system temperature, the licensee calculated a cold leg temperature applicable to each operating cycle and applied this value as representative of vessel head temperature for the EDY calculation. The licensee considered the cold leg temperature representative of operating head temperature because of the coolant bypass flow channels in the vessel head which allowed the inlet flow to the reactor vessel to pass along the vessel head (e.g., cold leg temperature). The licensee concluded that this design feature applied to the Braidwood Unit 1 vessel head based upon information in Westinghouse Letter CA-RPV-076, Confirmatory Measurement of Upper Head Temperature for Byron Class Plants, and WCAP 11444, T_{hot} Reduction Reactor Vessel Evaluation.

g. During non-visual examinations, was the disposition of indications consistent with the guidance provided in Appendix D of this TI? If not, was a more restrictive flaw evaluation guidance used?

Not applicable. The licensee did not identify any indications for which required a flaw evaluation.

h. Did procedures exist to identify potential boric acid leaks from pressure-retaining components above the vessel head?

Yes. Procedure ER-AP-331-1001, BACC Inspection Locations, Implementation and Inspection Guidelines, contained general walkdown inspection requirements. This procedure required BACC inspections after plant shutdown during each scheduled refueling outage by VT-2 examiners. To meet the requirements of NRC Order EA-03-009, the licensee performed a direct visual inspection of the RVCH through access doors in the service structure in accordance with this

procedure. The licensee did not identify any boric acid leaks from pressure-retaining components above the vessel head during this inspection.

i. Did the licensee perform appropriate follow-on examinations for boric acid leaks from pressure retaining components above the vessel head?

Not applicable. The licensee did not identify any boric acid leaks from pressure retaining components above the vessel head during the current refueling outage.

c. Findings

No findings of significance were identified.

.2 Operational Readiness of Offsite Power and Impact on Plant Risk (TI 2515/165)

This TI was completed and documented in Inspection Report 05000456/2006002; 05000457/2006002, Section 4OA5. During this inspection period the NRC requested followup information regarding the status of one operating procedure that was still in draft during the last inspection. The inspectors determined that the procedure had been issued on April 7, 2006.

.3 World Association of Nuclear Operators Peer Review Report Review

The inspectors and the NRC Branch Chief reviewed the final report, dated June 6, 2006, for the World Association of Nuclear Operators plant assessment conducted in December 2005.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. K. Polson and other members of licensee management at the conclusion of the inspection on June 30, 2006. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit was conducted for TI 2515/150, and ISI activities with Mr. K. Polson and other members of licensee management at the conclusion of the inspection on April 27, 2006. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

An interim exit meeting was conducted for the access control to radiologically significant areas program and the ALARA planning and controls program with Mr. J. Moser on April 28, 2006.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
K. Polson, Site Vice President
G. Boerschig, Plant Manager
D. Ambler, Regulatory Assurance Manager
G. Bal, Engineering Programs Manager
B. Casey, Engineering Programs, ISI
M. Cichon, Licensing Engineer
T. D'Antonio, Project Manager
H. Do, Engineering Programs, Corporate
G. Dudek, Operations Director
J. Gosnell, Tritium Team
A. Haeger, Tritium Team
J. Moser, Radiation Protection Manager
M. Sears, Steam Generator Program Manager
M. Smith, Engineering Director

Nuclear Regulatory Commission
R. Skokowski, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000456/2006003-01; 05000457/2006003-01	NCV	Failure to maintain fire barrier in accordance with fire protection program (Section 1R05)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection
Work Order 819986-01; High Temperature Equipment Protection
IR 337195; Summer Readiness Lesson Learned: Revise Bus Duct Fans Preventive Maintenance; May 20, 2005
IR 347779; Rescheduling of Action Tracking Item Affects Summer Readiness Issue; June 27, 2005
IR 383327; Potential Trend in Summer Readiness Issues; October 7, 2005
IR 479318; Seasonal Readiness Peer Review Results; April 16, 2006
WC-AA-107; Seasonal Readiness; Revision 2

1R04 Equipment Alignment
BwOP CS-E2; Electrical Lineup - Unit 2 Containment Spray System Electrical Lineup; Revision OE2
BwOP CS-M2; Operating Mechanical Lineup Unit 2; Revision 6
BwOP DG-E3; Electrical Lineup - Unit 2 2A DG; Revision 5
BwOP DG-M3; Operating Mechanical Lineup Unit 2 2A DG; Revision 12
BwOP RH-E1; Electrical Lineup - Unit 1 Operating; Revision 6
BwOP RH-M1; Operating Mechanical Lineup Unit 1 1A RH Train; Revision 12
IR 433300; Pin Not fully Secure on Racking Screw of 2AP06EP; December 13, 2005

1R05 Fire Protection
Braidwood Station Individual Plant Examination of External Events Submittal Report; June 1997
Braidwood Station Pre-Fire Plans; Suppression Zone 2S-43 (Fire Zone 3.2A-2)
Braidwood Station Pre-fire Plans; Detection Zone 2D-49, 2D-50 (Fire Zone 3.2A-2)
Fire Protection Report 2.3.3.4; Unit 1 Nonsegregated Bus Duct Area (Fire Zone 3.2A-1); Amendment 18
Fire Protection Report 2.3.5.1; Division 12 ESF Switchgear room (Fire Area 5.1.1); Amendment 19
Fire Protection Report 2.3.5.4; division 21 ESF Switchgear room (Fire Area 5.2-2); Amendment 18
Fire Protection Report 2.3.9.2; DG Room 2B (Fire Zone 9.1.2); Amendment 18
Fire Protection Report 2.3.9.6; DG Day Tank room 2A (Fire Area 9.3-2); Amendment 15
Fire Protection Report 2.3.9.8; DG Day Tank room 2B (Fire Area 9.4-2); Amendment 15
Fire Protection Report; Figure 2.3-9, Lower Cable Spreading Room Elevation 439'-0"; Sheet 1 of 1
Fire Protection Report; Figure 2.3-10; Mezzanine Floor Plan 416'-0"; Sheet 1 of 4
Fire Protection Report; Figure 2.3-10; Mezzanine Floor Plan 416'-0"; Sheet 3 of 4
Fire Protection Report; Figure 2.3-12; Grade Floor 401'-0"; Sheet 3 of 4
Fire Protection Report; Figure 2.3-13; Plan at 383'-0"; Sheet 2 of 2
Fire Protection Report; Figure 2.3.36; Cable Tray Installation; 401'-0"; Sheet 1 of 2

BwAP 1110-1A3; GOCAR Required compensatory Measures Action Response Fire Protection Water Suppression Systems; Revision 5
IR 472351; NRC Questions Temporary Storage Inside Steam Generator Replacement Project Building; March 29, 2006 [NRC-Identified]
IR 477902; Fire Dampers Not Installed in Fire Rated Barrier; April 12, 2006 [NRC-Identified]
IR 481655; Poor Housekeeping Identified in Lower cable Spreading Room Zone 2Z1; April 21, 2006 [NRC-Identified]
IR 482093; Inspect and Clean All Cable Pans in Lower Cable Spreading Room; April 21, 2006 [NRC-Identified]
IR 494921; Questions Regarding D-337 and Compliance With Plant Barrier Impairment Requirements; May 30, 2006 [IEMA-Identified]
IR 497885; Sprinkler Head Dripping Water on floor of 2B Diesel Oil Storage Tank Room; June 8, 2006
Braidwood Station's Generic Letter 86-10 Evaluation; Fire Protection Evaluation for Fire Zones 11.3-1/12.1-0 and 11.3-2/12.1-0 Boundaries to Demonstrate Separation Equivalent to Branch Technical Position CMEB 9.5-1, C5.b(2)

1R06 Flood Protection Measures
OP-AA-108-111-1001; Severe Weather and Natural Disaster Guidelines; Revision 2
Braidwood Station Individual Plant Examination of External Events Submittal Report; June 1997; Page 5-5
UFSAR Section 3.4 Water Level (Flood) Design; Revision 10, December 2004
IR 478006; Improper Blockage of West side Drainage Ditches; April 6, 2006 [NRC-Identified]

1R07 Heat Sink Performance
BwVS 900-29; Heat Transfer Test for Component Cooling Water Heat Exchangers 1CC01A; Revision 8
Braidwood - All 3 Component Cooling Heat Exchangers; Design and Corrected Overall Heat Transfer Capacity vs. Time Graphs

1R08 Inservice Inspection Activities

Boric Acid Related Corrective Action Program Documents and Evaluations
Attachment 1; Identification of Boric Acid Leakage Unit 1 Excess Letdown Heat Exchanger 1CV01AB-1B & 1CV01AA-1A; April 30, 2003
Attachment 3; Evaluation of Leakage from Bolted Connection Unit 1 Excess Letdown Heat Exchanger 1CV01AA-1A; April 20, 2004
IR 215922; Permanent Repair of Excess Letdown Heat Exchanger Deferred to A1R-12; April 20, 2004
IR 327428; Boric Acid on Orifice Connection 2S104MC During Safety Injection System Test; April 20, 2005
IR 335656; Boric Acid Accumulation Found in 383' Pipe Tunnel; May 16, 2005
IR 335683; Components on Surfaces Affected by Leakage from Filter 1CV02; May 16, 2005
IR 344845; 2B Residual Heat Removal Heat Exchanger; June 13, 2005
IR 480489; Boric Acid Accumulation at Bottom of Pressurizer; April 19, 2006

Corrective Action Program Documents
IR 310602; Small Thru-Wall Leakage on 1A Feedwater Pump Casing; March 9, 2005

IR 329076; Foreign Objects Found 2A/2D SG; April 26, 2005
IR 329131; Foreign Object Found 2C SG at TSP 08H; April 27, 2005
IR 331690; Thru-Wall Leakage on Primary System Piping; May 4, 2005
IR 338480; ASME Pressure Test Frequency not met; May 25, 2005
IR 385066; 2A Residual Heat Removal Heat Exchanger Inlet Flange Corrosion; October 12, 2005
IR 385077; 2A Residual Heat Removal heat Exchanger Outlet Flange Corrosion; October 12, 2005

Documents Related to Code Pressure Boundary Welding
Work Order 00609939-01;1SI-8819D - Cutout and Replace 2" Kerotest Check Valve; October 14, 2004
Liquid Penetrant Examination Data Sheets; FW-6 and FW 25; September 24, 2004
Weld Procedure Specification; 8-8 GTSM Manual GTAW, SMAW; Revision 1
Qualification Record 1-51A; December 28, 1983
Qualification Record 4-51A; April 20, 2001
Qualification Record A-003; February 8, 2000
Qualification Record A-004; February 8, 2000

Corrective Action Documents As A Result of NRC Inspection
IR 482126; Issues With Data Acquisition of Control Rod Drive Mechanism Penetration #68; April 22, 2006 [NRC-Identified]
IR 482199; NRC Identified Issues with 1CV01AA-1A and 1CV01AB-1B ASME Evaluations; April 23, 2006 [NRC-Identified]
IR 482394; NRC Identified Errors in BACC Evaluations; April 24, 2006 [NRC-Identified]
IR 483156; Evaluate Temporary Scaffold for Permanent Installation; April 25, 2006 [NRC-Identified]
IR 483826; A1R12 NRC Order EA-03-009 Interpretation Discrepancy; April 26, 2006 [NRC-Identified]

Documents Associated with ASME Code NDE
EXE-PDI-UT-2; Ultrasonic Examination of Austenitic Piping Welds in Accordance with PDI-UT-2; Revision 4
EXE-PDI-UT-1; Ultrasonic Examination of Ferritic Piping Welds in Accordance with PDI-UT-1; Revision 4
ER-AA-335-031; Ultrasonic Examination of Austenitic Piping Welds; Revision 2
ER-AA-335-030; Ultrasonic Examination of Ferritic Piping Welds; Revision 2
ER-AA-335-002; Liquid Penetrant Examination; Revision 3
EXE-ISI-11; Liquid Penetrant Examination; Revision 1

Documents Associated with Disposition of Relevant Indications
Enclosure -1 VT2, Visual Examination Record; Leakage from Heat Exchanger 1CV01AA and 1CV01AB; April 30, 2003
IR 215922; Permanent Repair of Excess Letdown Heat Exchanger Deferred to A1R-12; April 20, 2004

Other Documents
Braidwood Unit 1 A1R11 Condition Monitoring and Operational Assessment Report; Revision 0

Braidwood Unit 1 SG Inspection Degradation Assessment and Condition Monitoring Input Checklist for A1R12; January 17, 2006
ER-MW-335-1009; Site Specific Performance Demonstration Program; Revision 2
ER-AP-335-040; Evaluation of Eddy Current Data for SG Tubing; Revision 2
ER-AP-420-006; Byron/Braidwood Unit 1: SG Secondary Side Visual Surveillance Activities; Revision 3
SG-SGDA-06-06; SG Foreign Object Wear Scar Structural Limit Evaluation; April 6, 2006
Westinghouse Document; Use of Appendix H Qualified Techniques at Braidwood A1R12 Outage; March 6, 2006
Westinghouse Document MRS 2.4.2- GEN 45; Standard In-Situ Pressure Test Using the Computerized Data Acquisition Equipment Exelon Nuclear Power Plants; Revision 5

1R11 Licensed Operator Requalification Program
Out of the Box Scenario 0631; Pressurizer Pressure Failure/ Technical Specification Action Ramp/High Reactor Coolant System Activity

1R12 Maintenance Effectiveness
IR 223097; 0VC05YD Hydrometer Did Not Respond to an Open signal; May 24, 2004
IR 301555; Per IR 223097 0FZ-VC024E and 0FZ-VC024F Need to be Replaced; February 15, 2005
IR 302231; Breaker F4 Has Hot Spot on Top Left Lug of Breaker; February 16, 2005
IR 304273; Shelf Life Extension Test Results for Charcoal Acceptance; February 22, 2005
IR 305079; Elevated Motor Temperature of the 0VC13J Vent Fan; February 24, 2005
IR 306471; Low Oil Level Forces Shutdown of 0B VC Chiller; February 28, 2005
IR 306781; Various Concerns with Replacement Charcoal Inventory; March 1, 2005
IR 308442; Concern with Incomplete EC [Engineering Change] 348271; March 4, 2005
IR 317982; OPEX NER BY-05-017 Applicability to Braidwood; March 28, 2005
IR 334291; High Vibrations on 2MP02C After Maintenance; May 11, 2005
IR 337527; 1MP01C - Vibration Traces Indicate Degraded Motor Bearings; May 23, 2005
IR 340716; High Vibration on Fan Bearings on 1MP02C; June 3, 2005
IR 341166; Time Delay Relays 2-1 and 2-2 Failed; June 5, 2005
IR 349315; 2W MPT High Oil Temperature; June 30, 2005
IR 369986; 0VC02CA Indications of Breaker Cycling During Train Swap; September 4, 2005
IR 399042; Green NRC Finding From Safety System Design Inspection; November 15, 2005
IR 429549; Debris Noted In Transformer Yards Affecting MPT Cooling, December 1, 2005
IR 430142; Exciter Damage on 2MP09E; December 3, 2005
IR 432486; Post Maintenance Testing Delay VC Limiting Condition for Operation Exit; December 9, 2005
IR 435876; Review of OE21839 - Main Generator Load Reject and Reactor Trip; December 21, 2005
IR 439938; Need To Evaluate Bus Duct Cooling Alarm System; January 8, 2006
IR 444862; Adverse Trend On Main Power Transformer Deficiencies; January 23, 2006
IR 445838; Potential Foreign Material In Iso-Phase Bus Duct; January 25, 2006
IR 469919; Loss of U2 Main Gen Ground Detection; March 23, 2006
IR 470214; Potential Trend with VC Chiller Problems; March 24, 2006
IR 475712; Obtain Fuses for Analysis from 0VC01JA; April 6, 2006
IR 475716; Obtain Fuses for Analysis from OVC01JB; April 6, 2006
IR 476338; 0WO01CA - 'A' Train control Room chiller Failed to Start due to Wiring Error; April 3, 2006

IR 476744; Difference Between Prints and Infield Fuse Size/Type; April 10, 2006
IR478765; Adverse Trend For Total Dissolved Combustible Gas on MPT 1E; April 14, 2006
IR 485657; Operating Experience Review of GE SGC21 Negative Phase Sequence Relays; May 2, 2006
IR 486580; Terminal Blocks On The Control Panel Are Corroded, May 4, 2006
IR 486834; Unit 1 Main Generator Field Brush Failure Alarm; May 4, 2006
IR 487598; Unexpected U-2 Auto Ground Detector Operation; May 7, 2006
IR 494430; 1E MPT Cooling Coils Need To Be Cleaned; May 28, 2006
IR 494635; Unit 2 Generator Condition Monitor Does Not Generate Alarms; May 30, 2006
IR 497990; Discrepancies During Preventive Maintenance On 4160 Volt Breaker Removed From 2AP06EQ; June 8, 2006
IR 498614; Flexible Line Showing Signs Of Deterioration; June 9, 2006
IR 501482; BT9-15 "B" Phase Bus 15 Side Elevated Temp Of Doble Test Tap; June 19, 2006
WO 00925946; Trouble Alarm Received On Air Circuit Breaker 14-15 During Thunderstorm; May 31, 2006
Braidwood's Archival Operations Narrative Logs; February 1,2004 and Before December 31, 2005
BwOP MP-14; Generator Condition Monitor Operation; Revision 6
Drawing —96; Diagram of Control Room Heating Ventilation and Air Conditioning System; Sheets 1, 2, 3, and 4 of 4
VC-1 Control Room Vent Drawing; August 12, 2005; Revision 8
Maintenance Rule - Evaluation History; MR System MP
Maintenance Rule - Performance Criteria; System MP
Maintenance Rule - High Safety Significant Status Of In-Scope Functions; MR System MP
Maintenance Rule - Expert Panel Scoping Determination; System MP
Maintenance Rule - Evaluation History; MR System VC
Maintenance Rule - Performance Criteria; System VC
ATI 381794-15; Risk Analysis for VC Fuse Failure Concern; April 5, 2006
AC-1, Switchyard, Revision 7, May 23, 2006
AC-6, Alternating Current Power Notes, Revision 10, November 30, 2005

1R13 Maintenance Risk Assessments and Emergent Work Control
2BwOSR 3.8.1.1; Unit 2 Offsite AC Power Availability Surveillance; Revision 3
IR 491282; Protected Equipment Sign Left in Place After Work Complete; May 18, 2006 [NRC-Identified]
1RH01PB Protected Equipment
2B DG (2DG01KB) Protected Equipment; June 11, 2006
Unit 0, 1 Risk Assessment; May 15, 2006
OU-AA-103; Shutdown Safety Management Program; Revision 6
OU-AA-103; Shutdown Safety Approval; Attachment 1; Revision 6
OU-AP-104; Vital Systems; Attachment 8; Revision 8
OU-AP-104; Manual Safety Assessment Flowcharts Configuration 8 [Mode 4]: Vital Systems; Revision 8
Drawing —42; Diagram of Essential Service Water Units 1 and 2; Sheets 1A and 1B
Unit 0, 1 Risk Assessment; 0CC01A Oot of Service Protected Equipment; May 30, 2006
Unit 2 Risk Assessment; 0CC01A Out of Service Protected Equipment; May 30, 2006
Unit 0 Component Cooling Heat Exchanger Protected Equipment Log; May 30, 2006

1R14 Operator Performance During Non-Routine Evolutions and Events
IR 475790; 25B Drain Cooler Shell Side Relief Lifted; April 6, 2006
IR 476165; Safety Near Miss - 25B Drain Cooler Relief Valve Failure; April 7, 2006
IR 478006; Improper Blockage of West Side Drainage Ditches; April 12, 2006 [NRC-Identified]
IR 478670; Lessons Learned from 25B Drain Cooler Relief Valve Failure; April 14, 2006
Tritium Sample Data Sheets for site and ditch samples on April 6-7, 2006
Braidwood Archival Operations Narrative Logs for April 6, 2006
Braidwood UFSAR Section 2.4.2.3; Effects of Local Intense Precipitation

1R15 Operability Evaluations
IR 076282; 1VP01CB Broken Turning Vane; September 23, 2001
IR 156091; 1VP01CB - Turning Vane Materials Not to Desired Strength; April 28, 2003
IR 185469; 2C RCFC Turning Vane Degraded (Piece Detached); November 8, 2003
IR 232417; 1D RCFC Flow Less than TS Limit and Limiting Condition for Operations Entry; June 29, 2004
IR 261392; 1C RCFC Fan Turning Vane Found Cracked; October 7, 2004
IR 262227; Crack Next to Weld on 1VP01CB; October 11, 2004
IR 265273; RCFC Turning Vane Difference From Byron's Design; October 20, 2004
IR 326978; 2B RCFC Turning Vane Has 3 Cracks; April 21, 2005
IR 328095; Crack in 2C RCFC Turning Vane; April 24, 2005
IR 330397; NRC Concerns With Cracked Turning Vanes on RCFC's; April 29, 2005
IR 330397 #02; Apparent Cause Evaluation; Cracking of the Largest of 3 Turning Vanes Located in Elbow at Discharge of RCFC Fans; July 1, 2005
IR 482208; 1A RCFC Turning Vane Crack; April 23, 2006
IR 482848; 1C RCFC Turning Vane Crack; April 25, 2006
IR 496992; Spent Fuel Pool High Water Temperature; June 6, 2006
IR 500602; Fuel Pool Temperature at 101 with Rounds Maximum of 100 Degrees; June 16, 2006
MA-AA-796-024; Scaffold Installation, Inspection, and Removal; Revision 4
MA-MW-796-101; Attachment 2 Exhibit G-AWS Weld Man 1VP01CP; Revision 2
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WDI-TJ-1010; Evaluation of the Effect of Changing Secondary Frequency Used for J-Weld Inspection; Revision 0
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Wesdyne Procedure WPD 9.2; Qualification and Certification of Personnel in Nondestructive Examination; Revision 3
Westinghouse Letter CA-RPV-076; Confirmatory Measurement of Upper Head Temperature for Byron Class Plants; April 12, 1982

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ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
BwAP	Braidwood Administrative Procedure
BwOP	Braidwood Operating Procedure
BwOSR	Braidwood Operating Surveillance Requirement Procedure
BwVS	Braidwood Engineering Surveillance Procedure
BwVSR	Braidwood Engineering Surveillance Requirement Procedure
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CS	Containment Spray
DG	Diesel Generator
EC	Engineering Change
EDY	Effective Degradation Years
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ET	Eddy Current Examination
FRAC	Fixed Rear Axil Vehicle
IR	Issue Reports
ISI	Inservice Inspection
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PWSCC	Primary Water Stress Corrosion Cracking
RCFC	Reactor Containment Fan Cooler
RH	Residual Heat Removal
RVCH	Reactor Vessel Closure Head
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Steam Generator
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VHP	Vessel Head Penetration
VT	Visual Examination